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November 15, 2017
L-17-317

10 CFR 50.55a(z)(2)

ATTN: Document Control Desk
U. S. Nuclear Regulatory Commission
Washington, DC 20555-0001

SUBJECT:
Beaver Valley Power Station, Unit No. 1
Docket No. 50-334, License No. DPR-66
Request to Extend Certain Reactor Vessel Inspections From 10 to 20 Years (Request 1-TYP-4-BN-01)

In accordance with the provisions of 10 CFR 50.55a(z), FirstEnergy Nuclear Operating Company (FENOC) hereby requests Nuclear Regulatory Commission (NRC) approval of proposed alternative to American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME BPV Code), Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," subsubarticle IWA-2430, "Inspection Intervals," and Table IWB-2500-1, "Examination Categories," at Beaver Valley Power Station, Unit No. 1.

The proposed alternative would extend the inservice inspection interval from 10 to 20 years for certain reactor vessel interior attachment welds and core support structure surfaces. A more detailed description of the proposed alternative and supporting information is enclosed.

FENOC requests approval of the proposed alternative by March 30, 2018 to support the spring 2018 maintenance and refueling outage. There are no regulatory commitments contained in this submittal. If there are any questions or if additional information is required, please contact Mr. Thomas A. Lentz, Manager – Fleet Licensing, at 330-315-6810.

Sincerely,

Richard D. Bologna

Enclosure:

Beaver Valley Power Station, Unit No. 1,
10 CFR 50.55a Request 1-TYP-4-BN-01, Revision 0

cc: NRC Region I Administrator
NRC Resident Inspector
NRC Project Manager
Director BRP/DEP
Site BRP/DEP Representative

Enclosure
L-17-317

Beaver Valley Power Station, Unit No. 1,
10 CFR 50.55a Request 1-TYP-4-BN-01, Revision 0
(12 Pages Follow)

Proposed Alternative
In Accordance with 10 CFR 50.55a(z)(2)

- Hardship Without A Compensating Increase in Quality and Safety -

1. ASME Code Component(s) Affected

The affected component is the Beaver Valley Power Station, Unit No. 1 (BVPS-1) reactor vessel, specifically, the reactor vessel sub-components listed below. The applicable American Society of Mechanical Engineers, Boiler and Pressure Vessel Code (ASME BPV Code), Section XI, "Rules for Inservice Inspection of Nuclear Power Plant Components," examination categories and item numbers are also listed. The listed examination categories (hereafter referred to as category) and item numbers are from Subarticle IWB-2500 and Table IWB-2500-1 of ASME BPV Code, Section XI.

Examination Category	Item No.	Description
B-N-2	B13.60	Interior Attachments Beyond Beltline Region
B-N-3	B13.70	Core Support Structure

2. Applicable Code Edition and Addenda

ASME BPV Code, Section XI, 2001 Edition through the 2003 Addenda.

3. Applicable Code Requirement

Subsubarticle IWA-2430, "Inspection Intervals," subparagraph (d)(1) states that:

> Each inspection interval may be reduced or extended by as much as one year. Adjustments shall not cause successive intervals to be altered by more than one year from the original pattern of intervals. If an inspection interval is extended, neither the start and end dates nor the inservice inspection program for the successive interval need be revised.

Table IWB-2500-1, "Examination Categories," includes the following reactor pressure vessel visual examinations once each 10-year interval, among other examination requirements:

- Item number B13.60, accessible interior attachment welds beyond the beltline region (category B-N-2), and
- Item number B13.70, accessible core support structure surfaces (category B-N-3).

4. Reason for Request

The current BVPS-1 fourth 10-year inservice inspection interval began on April 1, 2008 and is scheduled to end on August 28, 2018. Request 1-TYP-4-BA-01 (Accession Number ML17297A318) proposes an alternative to the ASME BPV Code requirements pursuant to 10 CFR 50.55a(z)(1) on the basis that the extended inspection interval (20-year interval ending August 28, 2028) provides an acceptable level of quality and safety.

Request 1-TYP-4-BA-01 would permit future examination of the pressure-retaining reactor pressure vessel welds and full penetration reactor pressure vessel nozzle welds (category B-A and B-D reactor pressure vessel welds) to be performed during the maintenance and refueling outage currently scheduled in 2027. The core support structure must be removed to perform these examinations.

Request 1-TYP-4-BN-01 proposes to extend the fourth inspection interval for visual examination of the interior attachment welds beyond the reactor pressure vessel beltline region (category B-N-2), and visual examination of the accessible core support structure surfaces (category B-N-3) in order to allow deferral of the subject examinations to the same refueling outage as the category B-A and B-D reactor pressure vessel shell welds and nozzle welds described in request 1-TYP-4-BA-01. The core support structure must be removed to perform these examinations.

During the ten-year inservice inspection of the BVPS-1 reactor pressure vessel shell, lower head, and nozzle welds performed in 2007, FENOC also performed visual examinations of the accessible reactor pressure vessel interior attachment welds and the accessible core support structure surfaces. Since the core support structure requires removal to facilitate examination of the reactor pressure vessel shell, lower head, and nozzle welds, visual examinations of the reactor pressure vessel interior attachment welds and core support structure surfaces have historically been performed during the same outage at the end of the inservice inspection interval. Performing these related examinations during the same refueling outage results in significant savings in outage duration since the same equipment and personnel used for examination of the reactor pressure vessel shell, lower head, and nozzle welds from the reactor pressure vessel interior can implement the examinations of reactor pressure vessel interior attachment welds and core support structure surfaces.

The core support structure must be removed from the reactor vessel to perform the B-N-2 and B-N-3 examinations. The bottom of the core support structure is secured within the reactor vessel by clevis inserts/core support lugs that interface with radial keys on the outside diameter of the core barrel. Because of the close tolerance fit between the clevis inserts/core support lugs and the radial keys on the core barrel, precision lifts are required to remove and replace the core support structure for the B-N-2 and B-N-3 examinations. Every core support structure lift presents risk in that the reactor pressure vessel and reactor pressure vessel internals may be damaged during the lift.

In addition to the risk presented by the precision lifts required to remove and replace the core support structure, movement of the core support structure presents significant radiological risk. Because of the BVPS-1 refueling cavity design, the core support structure rises out of the water when it is moved between the reactor vessel and the storage stand. During movement of the radioactive core support structure in 2007, high radiation levels were observed in the BVPS-1 containment building, creating a risk of associated radiation exposure to workers. Removing the core support structure once instead of twice during the proposed 20-year inspection interval to perform the B-N-2 and B-N-3 examinations would maintain radiation exposure as low as reasonably achievable.

Performing category B-N-2 and B-N-3 examinations and the category B-A and B- examinations during the same refueling outage results in a significant savings in outage duration, radiation exposure, and avoidance of a lift of a heavy close-fit component that has the potential for inflicting damage to itself, and reactor vessel surfaces.

5. Proposed Alternative and Basis for Use

As an alternative to ASME BPV Code, Section XI, Subsubarticle IWA-2430 and Table IWB-2500-1, FENOC proposes to extend the fourth 10-year inservice inspection interval for the category B-N-2 reactor vessel interior attachment welds beyond the beltline region and the category B-N-3 reactor vessel core support structure surfaces until August 28, 2028. Currently, the subject examinations will need to be performed before the end of the Spring 2018 refueling outage. The proposed alternative inspection would enable the subject examinations to be performed during the refueling outage in 2027. This would align the subject examinations with the risk-informed extension of the inservice inspection interval for category B-A reactor vessel pressure-retaining welds and category B-D nozzle-to-vessel and nozzle inner radius section welds that was submitted in relief request 1-TYP-4-BA-01.

If there is an opportunity to perform examinations due to moving the core support structure from the reactor vessel to the permanent storage stand prior to the 2027 refueling outage, the code required VT-3 examinations will be performed on the core support structure and interior reactor vessel attachments in order to fulfill the exam requirements for the extended interval.

The B-N-2 visual examinations of the reactor pressure vessel interior attachment welds have been performed twice at BVPS-1. The B-N-3 visual examinations of the reactor pressure vessel core support structure have been performed three times at BVPS-1. There were no relevant conditions observed during the B-N-2 and B-N-3 examinations. During the 1986 B-N-3 examination, five indications were observed on the lower internals. The observed indications were surface related, and were identified as scraped areas, minor damage, and minor porosity. The five indications were assessed and found not to affect the structural integrity of the core support structure. The B-N-2 and B-N-3 examinations were last performed during the 2007 maintenance and refueling outage with no indications observed. Additionally, operating experience indicates that these examinations have been performed many times by the industry without any significant findings associated with components present in the BVPS-1 reactor vessel design.

During the 2016 BVPS-1 refueling outage, FENOC performed the category B-N-1 visual examination. This examination included the reactor vessel interior areas that are made accessible for examination by the removal of components during normal refueling outages. There were no relevant conditions observed during the 2016 B-N-1 visual examination. This examination is required once each inspection period.

The BVPS-1 reactor pressure vessel contains four core support lugs that were fabricated from Alloy 600 base material and attached to the reactor pressure vessel wall with alloy 82/182 welds. Because alloy 600/82/182 material is susceptible to primary water stress corrosion cracking (PWSCC), a PWSCC susceptibility assessment was

performed for the core support lugs. Based on low normal operating stresses, low normal operating temperature, a favorable operating environment, and the application of a post weld heat treatment, it is very unlikely that PWSCC will occur in the core support lugs and attachment welds during the operating life of BVPS-1. The core support lug PWSCC assessment is documented in the attached Westinghouse letter LTR-SDA-17-018, Revision 1, "Technical Basis for Beaver Valley Unit 1 Core Support Lug Examination Extension."

Reasonable assurance of structural integrity is provided by the following:

- There were no relevant conditions observed during the recent category B-N-1 visual examination that was performed during the 2016 BVPS-1 refueling outage.
- There were no relevant conditions observed during the B-N-2 and B-N-3 examinations that have been performed at BVPS-1.
- No significant findings have been observed in the industry associated with components present in the BVPS-1 reactor vessel design.
- The attached core support lug PWSCC-susceptibility assessment for BVPS-1.

This request would extend the duration of the third 10-year inspection interval for category B-N-2 and B-N-3, item numbers B13.60 and B13.70, visual examinations to 20 years. The proposed inspection interval extension is an alternative to subsubarticle IWA-2430 requirements regarding inspection interval length and Table IWB-2500-1 item numbers B13.60 and B13.70 requirements regarding examination frequency. Removing the core support structure for the sole purpose of performing the B-N-2 and B-N-3 examinations would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

6. Duration of Proposed Alternative

The proposed alternative would extend the duration of the fourth 10-year inspection interval for category B-N-2 and B-N-3, item numbers B13.60 and B13.70, visual examinations to August 28, 2028.

7. Precedent

A similar request was proposed for Vogtle Electric Generating Plant, Unit Nos. 1 and 2, that requested that the visual examinations for Category B-N-2 and B-N-3 components be performed consistent with the proposed inspection interval for Category B-A and B-D volumetric examinations. By letter dated June 27, 2017, (ADAMS Accession Number ML17171A102), the NRC staff authorized use of the alternative.

Attachment
10 CFR 50.55a Request 1-TYP-4-BN-01, Revision 0

(7 Pages Follow)

LTR-SDA-17-018, Revision 1
Technical Basis for Beaver Valley Unit 1 Core Support Lug Examination Extension

Westinghouse

To: Brendan Rodgers
cc: Steve McKinney, Tony Jurgovan

From: Stephen E. Marlette
Ext: (412) 374-3631
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Date: 11/15/17

Your ref: LTR-SDA-17-018, Rev. 1
Our ref: LTR-SDA-17-018, Rev. 1

Subject: Technical Basis for Beaver Valley Unit 1 Core Support Lug Examination Extension

References:

1. Combustion Engineering Report, CENC-1183, Rev. 0, "Analytical Report for Duquesne Light Company Beaver Valley Power Station Unit No. 1 Reactor Vessel," July 1972.
2. W. Bamford, G. White, S. Fyfitch, "Technical Basis for Stress Levels Needed to Mitigate PWSCC in Alloy 82/182/600," Proceedings of the ASME 2016 Pressure Vessels and Piping Conference, 2016, PVP2016-64041.
3. Combustion Engineering Drawing, E-233-714, Rev. 4, "Pressure Vessel Welding and Machining for: Westinghouse Electric Corporation 157 " I.D. P.W.R."
4. Combustion Engineering Drawing, A-244-001, Rev. 7, "Vessel Welding and Mach. 157" I.D. PWR."
5. Combustion Engineering Inc. Welding Procedure, SAA-701-1, Rev. 0, "SAA-701 Weld Procedures."
6. Westinghouse Report, WCAP-15829, Rev. 0, "Addendum to Analytical Report for Duquesne Light Company Beaver Valley Power Station Unit No. 1 Reactor Vessel (9.4% Power Uprating Evaluation)," February 2002.
7. P. Scott, et al., "Comparison of Laboratory and Field Experience of PWSCC in Alloy 182 Weld Metal," Proceedings of the 13th International Conference on Environmental Degradation of Materials in Nuclear Power Systems, paper 25, CNS, 2007.
8. Inconel Alloy 600, Publication SMC-027, Copyright ©Special Metals Corporation, 2008.
9. Materials Reliability Program: Pressurized Water Reactor Issue Management Tables – Revision 3 (MRP-205)
10. Combustion Engineering Inc. Welding Procedure, MA-1002-0, Rev 0, "Detail Welding Procedure," August 4, 1969.

Record of Revision

This revision was made to change the proprietary statement on pages 2 through 7 to "Non-Proprietary Class 3" from "Proprietary Class 3."

Background and Purpose

Beaver Valley Unit 1 is seeking to extend the B-N-2 and B-N-3 examination schedule from 10 years to 20 years for the reactor vessel core support lugs. The purpose of this letter report is to provide justification as to why this extension is acceptable. The core support lugs are made from Alloy 600 base metal and welded to the reactor vessel with Alloy 82/182 weld metal. Therefore, primary water stress corrosion cracking (PWSCC) would be the main concern for potential degradation during operation. PWSCC is attributed to three factors occurring simultaneously: environment (primary water), steady state operating temperature and residual plus operating stresses. In the case of the core support lugs, the low operating plus residual stresses and the low operating temperature significantly reduce the probability of crack initiation during operation for this component. Proof of this significant reduction in the probability of cracking is the excellent service experience of these core support lugs, industry-wide, with no cracking reported in over 40 years of service. The following paragraphs provide detailed discussion on various factors that create favorable conditions for the Beaver Valley Unit 1 core support lugs relative to PWSCC resistance and low probability of component failure.

Operating Stresses, and Threshold for Initiation

An ASME III analysis of the core support lugs was performed in the design basis stress report for Beaver Valley Unit 1 [1]. The core support lugs and attachment welds were evaluated against membrane plus bending stress and fatigue limits. The maximum design stress is the hoop stress with respect to the reactor vessel, which occurs at the outer periphery of the core support lug attachment weld. This stress is 24.93 ksi and is due to design pressure (2,500 psi). In comparison, the longitudinal stress is 11.85 ksi and the radial stress is -2.5 ksi (compressive) due to design pressure. The operating hoop stress would be slightly less considering normal operating pressure at 2,250 psi. Thus, the maximum normal operating hoop stress in the weld would be 22.44 ksi due to pressure. There is also a small additional membrane plus bending stress of 1.27 ksi within the weld due to a steady interface load, but no significant thermal stress at steady state operation. Therefore, the total maximum operating hoop stress in the weld is 23.71 ksi. The maximum normal operating principal stress within the core support lug base metal was calculated in [1] to be 10.45 ksi.

Since the concern is with the potential for PWSCC, all stresses which occur at steady state operation should be considered in this assessment. Stresses from insertion loads and cyclic stresses such as those due to vibration are not a concern for PWSCC, but were addressed in the design basis reports [1] and [6] to ensure that ASME Section III stress and fatigue limits were met. The only other stress of interest here is the residual stress from welding the core support lug to the vessel, but this stress has been minimized by post weld heat treatment (PWHT) of the core support lugs with the reactor vessel. This subject will be further discussed below.

As discussed in Reference [2], PWSCC initiation does not typically occur with stress levels below the yield strength of the material. A conservative stress threshold for crack initiation in Alloy 600 base metal was established in [2] at 20 ksi based on a minimum room temperature yield strength of 30 ksi for head penetration tubes. The yield strength of the Alloy 600, 82 and 182 at typical T_{hot} temperature (617 °F) is approximately 80 % of the room temperature yield strength. The estimated PWSCC initiation stress in [2] was established for head penetration tubes at 24 ksi, then reduced to 20 ksi (67% of yield) for

conservatism. Based on several publications discussed in [2], the room temperature yield strength for Alloy 82/182 is approximately 51 ksi, which is well above that of Alloy 600 base metal yield strength. Reference [8] also lists the room temperature yield strengths for the weld metals to be 55.1 ksi for Alloy 182 and 57.1 ksi for Alloy 82. Thus, 51 ksi is a conservative estimate of the room temperature yield strength for the Beaver Valley Unit 1 core support lug attachment welds. Using the same conservative approach used in [2], the PWSCC threshold would be closer to 34 to 40 ksi for the weld metal. Therefore, the normal operating stress of 23.71 ksi is well below the expected PWSCC threshold for the core support lug attachment welds and butter.

In order to provide an estimate of the improved performance at the low stress level within the core support welds, predictive models discussed in [2] provide a useful tool. Amzallag and others [2] have suggested that the time to PWSCC initiation in Alloy 600 base metal is proportional to the operating stress to the negative forth power. As discussed in [2], the following equation can also be conservatively applied to weld metal:

$$1/t = B\sigma^4 \qquad (1)$$

Experimental results discussed in [2] conservatively predict cracking of Alloy 182 at 730,000 hours (83 years) at 554 °F (290 °C), with an applied stress of the at-temperature yield strength (51 ksi). Considering Equation (1), the time to cracking for the core support lug weld can be conservatively predicted to occur at 1,784 years, using the normal operating stress of 23.71 ksi.

Likewise, the time to base metal cracking can be estimated considering the predictive model discussed in [2] for Alloy 600 base metal and the maximum operating stress reported in [1] for the Beaver Valley Unit 1 core support lugs. Reference [2] uses a model based on the predicted time to cracking in Alloy 600 base metal at 100,000 hours at the yield strength of the material (35 ksi), at 617 °F. Considering the maximum operating stress of 10.45 ksi, cracking in the base metal would not be predicted to occur until 1,436 years of operation at 617 °F.

Post Weld Heat Treatment

The Beaver Valley Unit 1 core support lug attachment weld and butter were heat treated after welding in accordance with fabrication documents [3], [4], [5] and [10]. Post weld heat treatment (PWHT) at 1,100 °F is performed to reduce, or relieve, the undesirable effects of welding within the ferritic steel base metal. Heat treatment also has beneficial effects on the PWSCC resistance of the Alloy 82/182 weld metal adjacent to the low alloy steel base metal. Reference [7] cites these benefits as being attributed to relief of weld residual surface stresses and change in microstructure of nickel alloy welds due to PWHT. Scott [7] states that PWHT results in recrystallization of the cold-worked surface layer of the weld due to grinding and significantly reduces residual surface stresses. Scott also notes in [7] that there has never been any observed cracking in reactor pressure vessel core radial supports, which are always stress relieved. Hence, the PWHT applied to the Beaver Valley Unit 1 core support lugs and butter provides improved resistance to PWSCC and has reduced or removed the residual stresses from the fabrication process.

Although the majority of the residual weld stresses are expected to have been removed, a conservative case can be considered for the weld material where the surface residual stress remains at 20 ksi for the weld metal. This would increase the normal operating plus residual stress to 43.71 ksi. Using this stress value and Equation 1 would reduce the predicted PWSCC initiation time to approximately 154 years of operation, which is still well beyond the operating life the plant.

Operating Temperature, and Predicted Time to Crack Initiation

The current maximum cold leg operating temperature for the Beaver Valley Unit 1 reactor vessel is 543.1 °F, which is within the low PWSCC susceptibility range for operating temperature. This was a slight decrease from the original inlet temperature of 543.5 °F, but the current operating range allows for inlet temperatures as low as 528.5 °F. However, for the purpose of evaluating susceptibility to PWSCC, the slightly higher original operating inlet temperature is conservatively used.

Reference [2] discusses a number of experiments that were conducted to determine the PWSCC initiation time for Alloy 600 base metal and Alloy 82/182 weld metals. The most limiting case is for the base metal, which is conservatively estimated to initiate cracks in 100,000 hours (11 years) at 617 °F (325 °C) based on experimental data. Considering the Beaver Valley core support lug original operating temperature of 543.5 °F, and using an activation energy of 185 kJ/mol with the Arrhenius equation, base metal cracking would not occur for at least 168 years. The base metal improvement to initiation time would be 15.3 times based on the lower operating temperature. Thus, the time to PWSCC initiation for the Beaver Valley Unit 1 support lug base metal would be extended to over 21,000 years when also considering the low operating stress discussed above.

Likewise, Alloy 182 was conservatively estimated to initiate cracking at 730,000 hours (83 years) at 554 °F (290 °C), with an applied stress at yield strength (51 ksi). Considering the Beaver Valley operating temperature (543.5 °F) alone would increase the time to cracking to approximately 126 years. Combining the time improvement due to the operating stress, the predicted time to PWSCC would be 2,698 years for the weld metal.

As previously discussed, PWHT is expected to have removed all or most of the residual stress within the core support lug welds, but a conservative case can be considered by adding 20 ksi to the operating stress to account for any remaining residual stress. Even in this extremely conservative case, the total time to initiation would be 234 years of operation considering the operating plus residual stress of 43.71 ksi and the inlet temperature of 543.5 °F.

Table 1 provides a summary of the results for time improvement due to the low operating stresses and temperature of the Beaver Valley Unit 1 core support lugs. As can be seen from the results, the predicted time to crack initiation is well beyond the operating life of the plant.

As noted above, cracking of the core support lugs or attachment welds has never been observed in any operating plant, and the Beaver Valley Unit 1 inlet temperature is lower than most in operation. Because operating temperature has a significant impact on the time to PWSCC initiation, PWSCC would more likely appear in other plants within the core support lug region before Beaver Valley Unit 1 as an early indicator within the industry.

Table 1 Comparison of Improvement in Time to PWSCC Initiation for Beaver Valley Unit 1 Support Lugs and Welds

Case	Stress (ksi)	Temp (°F)	Initiation Time w/Stress Improvement Only(2) (Years)	Initiation Time w/Temp Improvement Only(3) (Years)	Initiation Time w/Total Improvement(4) (Years)
BV1 Lug (A600)	10.45	543.5	1,436	168	21,960
BV1 Welds (A82/182)	23.71	543.5	1,784	126	2,698
BV1 Welds (A82/182)	43.71(1)	543.5	154	126	234

Notes:

1. Conservatively includes an assumed residual weld stress of 20 ksi within the core support lug welds even after PWHT
2. Calculated using Equation 1 considering improvement due to operating stress only
3. Calculated using the Arrhenius Equation considering improvement due to operating temperature only
4. Combined effect of lower operating stress and temperature

Operating Environment

The core support lugs are located in the down flow path of the PWR inlet water at T_{cold} temperatures and are below the beltline region of the vessel, so radiation effects are not significant. The flowing water prevents the concentration of impurity ions that could potentially accelerate degradation. There is a negligible amount of oxygen present, as the PWR environment is maintained with a hydrogen overpressure. As discussed above, the low operating temperature significantly reduces the potential for PWSCC, and there is no concern for radiation-assisted cracking due to the location of the lugs outside of the beltline region. Therefore, the operating environment of the core support lugs also reduces the likelihood of degradation during the operating life of the plant.

Likelihood and Consequences of Potential Cracking

In the unlikely event that cracking of the attachment weld or core support lug would occur, there would be no significant consequences in accordance with Table A-1 of MRP-205 [9] (ID numbers 1.5-4 and 1.5-5). The four core support lugs restrict the lateral and circumferential movement of the core barrel due to flow, vibration, and potential seismic events. There are two possible failure modes for cracking in the core support lugs. First, a crack could initiate and extend due to the maximum operating stress, which is the hoop stress relative to the vessel. A crack in this direction would be longitudinal, and would not affect the load carrying capacity of the lugs. Second, a crack could cause the lug to become detached from the reactor vessel, due to cracks propagating from the radial stresses or the shear stresses. However, as discussed above the radial stress is compressive and the shear stress is less than 1 ksi in the attachment weld. These stress levels are much lower than the hoop stresses in the weld, which means that the time to crack initiation would be much longer than those shown in Table 1.

It is also noted that two lugs would need to fail in shear at the same time to result in complete loss of lateral restraint. The calculated stresses within the core support lugs and welds due to normal operating loads are low compared to the material yield strengths of the base metal and welds (less than half) as reported in [1]. Consequently, a significant portion of the lug would need to be cracked before failure would occur. Therefore, loss of the core support lug restraint due to PWSCC is not a concern during operation.

Conclusion

Based on the discussions presented herein, it is very unlikely that PWSCC will occur in the core support lugs and attachment welds during the operating life of Beaver Valley Unit 1. The low normal operating stresses and temperatures of the core support lugs along with the constant flow of T_{cold} coolant across the lugs provide favorable conditions that significantly reduce the chance of PWSCC initiation. In addition, the PWHT applied to the lugs and welds during fabrication significantly reduces residual stresses, which are a significant contributor to PWSCC initiation. Furthermore, in the unlikely event that cracking would occur, complete loss of restraint from the core support lugs is not a concern since a significant degree of degradation would have to occur in two lugs at the same time. And finally, there has never been any PWSCC observed within the core support lugs or attachment welds in the history of PWR operation. Considering all of these factors, extension of the examination interval from 10 years to 20 years is acceptable for the Beaver Valley Unit 1 core support lugs.

If you have any questions, please contact the author at (412) 374-3631.

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